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International Agreement Report

Assessment of a PWR Control Rod Drop Transient with 3D Neutronic-Thermalhydraulic Coupled Codes RELAP5/PARCSv2.7 and TRACEv5.0P3/PARCSv3.0

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ABSTRACT

In the nuclear safety field, neutronic and thermal-hydraulic codes performance is an important issue. New capabilities implementation, models, and tools improvements are a significant part of the community effort in looking for better Nuclear Power Plants (NPP) designs.

A procedure to analyze the PWR response to local deviations on neutronic or thermal-hydraulic parameters is being developed. By neutronic-thermalhydraulic coupled codes, Incore and Excore neutron flux detector signals are simulated. These signals are compared, on the one hand, with the actual data collected during a control rod drop test at a PWR NPP and, on the other hand, with data obtained with SIMULATE-3K code, an advanced, two-group nodal code that delivers neutronic and thermal-hydraulic analysis with licensing-grade accuracy. At the same time, the used codes and their new capabilities are validated.

The 3D neutronic-thermalhydraulic codes used in this study are RELAP5/PARCSv2.7 and TRACEv5.0P3/PARCSv3.0. TRACEv5.0P3 and RELAP5 thermal-hydraulic models are full-core detailed models with three different azimuthal zones. Besides, the TRACE model is performed with a fully 3D core composed of a cartesian vessel representing the fuel assemblies and a cylindrical vessel representing the bypass and downcomer.

Cross-Section data are obtained from CASMO-4/SIMULATE-3 files using the SIMTAB methodology, which was developed at the Institute for Industrial, Radiophysical and Environmental Safety at Universitat Politècnica de València (ISIRYM-UPV) in collaboration with Iberdrola and has been validated for both PWR and BWR.

FOREWORD

This report represents one of the assessments or application calculations submitted to fulfill the bilateral agreement for cooperation in thermal-hydraulic activities between the Consejo de Seguridad Nuclear (CSN) and the U.S. Nuclear Regulatory Commission (NRC) in the form of a Spanish contribution to the NRC's Code Assessment and Management Program (CAMP), the main purpose of which is to validate the TRAC/RELAP Advanced Computational Engine (TRACE) code.

CSN, together with some relevant universities, have established a coordinated framework (CAMP-Spain) with two main objectives: to fulfill the formal CAMP requirements and to improve the quality of the technical support groups that provide services to the Spanish utilities, CSN, research centers, and engineering companies.

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EXECUTIVE SUMMARY

This work aims to study a Control Rod Drop Transient using coupled codes. During this control rod drop test done at a PWR nuclear power plant, the reactor was operating at 100% of its nominal power, and the insertion of a single rod was performed, releasing it from its locks and allowing its free fall. This control rod was chosen for its position in the reactor core: it was one of the rods with the most significant worth, i.e., the negative reactivity that a rod adds, and its situation about the detectors was suitable to the means of the test. A considerable amount of data was registered during this test. This data includes actual signals for Incore and Excore neutron detectors and their simulation results with SIMULATE-3K code [1].

To study the control rod drop transient, two thermalhydraulic-neutronic models were generated with the RELAP5/PARCSv2.7 and TRACEv5.0P3/PARCSv3.0 coupled codes.

PARCS neutronic model is a reactor core model. The core model is formed by 241 radial nodes and is divided into 34 axial levels. For each fuel assembly a radial node is assigned, so the active zone of the core is composed of 177 nodes, while 64 nodes are for the radial reflector. Between the 34 axial levels, 2 of them correspond to the top and bottom reflectors.

The thermal-hydraulic model generated with the RELAP5 code is a reactor core model. In contrast, the TRACE thermal-hydraulic model is a 3D reactor pressure vessel model, which allows the modelling of the cross-flow between each fuel assembly node.

In RELAP5, each fuel assembly is modeled with a *pipe+heat structure* component (following a one-to-one channel basis). The fuel assemblies, which represent the active zone of the core, are divided into 32 axial levels. The same set is used to model the bypass channels. In TRACE, the model consists of a cartesian vessel used to model the fuel assemblies together with a cylindrical vessel composed of two radial cells, three azimuthal sectors, and 36 axial cells. The inner radial cell represents the bypass, and the outer radial cell represents the downcomer. The three azimuthal cells represent the three different reactor recirculation loops. The lower and the upper axial cells represent the lower and upper reactor plenums. Setting to zero the flow area fraction for the corner cells, the cartesian vessel is shaped as the NPP radial mapping.

PARCS is coupled with RELAP5 and TRACE. These codes will feed PARCS with temperature and density distribution information during a transient. To perform the coupling, an input file called *MAPTAB*, is needed. It indicates the neutronic nodes assignment to each thermal-hydraulic node. It is obtained automatically thanks to the tools developed by ISIRYM-UPV with MATLAB® software [2].

Different versions of PARCS have been used. On the one hand, PARCSv3.0 is officially distributed and coupled with TRACE. On the other hand, RELAP5 is not provided with an official neutronic coupling code; therefore, due to previous experience with RELAP5/PARCSv2.7 couple code [3], [4],[5], it has been decided to continue with it, besides, the solver used in both versions of PARCS versions is the same.

Simulation of the detector signals from Incore and Excore detectors has been implemented in the model by changing the source code of PARCS. The implemented detector response is in accordance with the plant control logic features and characteristics. In this way, the code allows to search for the detector's position nodes and saves their main variables in separate files, with an acceptable format to optimize the analysis of the results [6].

The study of this PWR Control Rod Drop Transient concludes with an analysis of the signals of the detectors closest to the control rod dropped. The signal analysis of the detectors is performed by comparing the actual signals and the SIMULATE-3K results provided by CNAT with the results obtained by the coupled codes.

This study assesses the capacity of coupled codes RELAP5mod3.3/PARCSv2.7 and TRACEv5.0P3/PARCSv3.0 to replicate operational transients, such as control rod drop in a PWR NPP.

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The authors would like to acknowledge the economic support provided by CNAT and IBERDROLA, as well as their valuable collaboration and their willingness to develop this work. This work has also been supported by the Spanish Ministerio de Economía y Competitividad, through the projects NUC-MULTPHYS (ENE2012-34585) and VALIUN-3D (ENE2011-22823), and the Generalitat Valenciana (GVA), through the project PROMETEO II/2014/008.

ABBREVIATIONS AND ACRONYMS

ANM	Analytical Nodal Method
AOO	Anticipated Operational Occurrence
BOC	Beginning of the Cycle
CHF	Critical Heat Flux
CNAT	Almaraz-Trillo NPPs AIE
CPU	Central Processing Unit
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
IB	Iberdrola
ISIRYM	Institute for Industrial, Radiophysical and Environmental
NEM	Safety Nodal Expansion Method
NPP	Nuclear Power Plant
PARCS	Purdue Advanced Reactor Core Simulator
PDD	Power Density Detector
PVM	Parallel Virtual Machine
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RELAP5	Reaction Excursion and Leak Analysis Program
RIA	Reactivity Initiated Accidents
SCRAM	Safety Control Rod Axe Man (Emergency Reactor Shutdown)
SNAP	Symbolic Nuclear Analysis Package
TRACE	TRAC/RELAP Advanced Computational Engine
UPV	Universitat Politècnica de València
U.S. NRC	United States Nuclear Regulatory Commission

1 INTRODUCTION

A Nuclear Power Plant is designed not only to operate under nominal conditions, but to successfully withstand changes without undermining the reactor safety. These changes, that may be global or local, result from discrepancies between abnormal operating conditions and the expected normal conditions. It is interesting to develop tools to obtain the core local response to anticipated or postulated transients and accidents.

In this work, detailed thermalhydraulic-neutronic models of a PWR reactor core have been developed to perform an Incore and Excore neutron detector signal analysis. The signals from the detectors are analyzed in a transient scenario where a control rod drop test is performed. A control rod drop transient in a PWR is part of the Reactivity Initiated Accidents (RIA) in an NPP. There is vast literature regarding the analysis of these accidents [3], [4], [7].

Specifically, a control rod drop transient consists of the accidental insertion of a control rod due to a malfunction of its activation mechanism. A sudden and continuous negative reactivity insertion dominates the reactor power evolution, and the core power distribution is modified. Due to this absorber insertion, the reactivity decreases (as the neutron population decreases), and the reactor core turns subcritical. The effect of the moderator density and the Doppler temperature on the reactivity leads the reactor again to criticality in a few seconds, subsequently evolving into an asymptotic point.

The thermal-hydraulic evolution has little impact on the transient if the energy deposition is sufficiently low during this process to avoid a departure from nucleate boiling (DNB) [8]. However, it is necessary to know the specific acceptance criteria to analyze the proposed transient properly.

NUREG-0800, in chapter 15 [9], defines the acceptance criteria for Anticipated Operational Occurrences (AOOs). The acceptance criteria for AOOs, defined by the U. S. Nuclear Regulatory Commission, are established according to the following points:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 % of the design values by the ASME Boiler and Pressure Vessel Code.
2. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs.
3. An AOO should not generate a postulated accident without other faults occurring independently or resulting in a consequential loss of function of the Reactor Coolant System (RCS) or reactor containment barriers.

By definition, an AOO cannot generate an accident without other incidents occurring independently or resulting in a consequential loss of function of the RCS or reactor containment barriers. Therefore, point 3 is assured.

This study is based on a control rod drop test at a PWR NPP. During this test, the reactor was operating at 100% of its nominal power, and the insertion of a single rod was performed, releasing it from its locks and allowing its free fall. This control rod was chosen for its position in the reactor core: it was one of the rods with the most significant worth, i.e., the negative reactivity that a rod adds, and its situation about the detectors was suitable to the means of the test. A considerable amount of data was registered during this test. This data includes real signals for *Incore* and *Excore* neutron detectors and their simulation results with the SIMULATE-3K code. The test was carried out to assess a modification in the control logic that detects a control rod drop and was not related to the NPP safety analysis but has been very useful for this work. Since, besides analyzing the control rod drop transient from the safety point of view, the deepest analysis can be performed thanks to the actual data provided by Centrales Nucleares Almaraz-Trillo (CNAT) [1].

Different versions of PARCS have been used. On the one hand, PARCSv3.0 is officially distributed and coupled with TRACE. On the other hand, RELAP5 is not provided with an official neutronic coupling code; therefore, due to previous experience with RELAP5/PARCSv2.7 couple code, it has been decided to continue with it, besides, the solver used in both versions of PARCS versions is the same. Thus, this study presents a capacity test of RELAP5mod3.3/PARCSv2.7, and TRACEv5.0P3/PARCSv3.0 coupled codes to reproduce operational transients, such as a control rod drop in a PWR NPP.

2 PLANT DESCRIPTION

The Nuclear Power Plant studied is a pressurized water reactor with three cooling loops and a German Siemens-KWU design (KWU-PWR). Commercial operation started in 1988 and has a nominal reactor power of 3010 MWth.

The Reactor Pressure Vessel (RPV) is the fixed point of the reactor coolant loops. The three main cooling loops depart from the reactor vessel with an angle of 114º between them. Inside the RPV, the reactor core is housed. The reactor core layout consists of 177 fuel assemblies, each containing 236 fuel rods in a 16x16 square array. Each fuel rod is formed by the cylindrical Zircaloy cladding finished with two closure plugs, which houses the UO2 pickup stack. The 20 bars of each of the 52 control elements are inserted into the same number of fuel assemblies using the actuation coils located above the lid of the vessel. The control rods consist of neutron absorbent material (Ag, In, Cd) housed in pods whose lower plugs are inserted into the guide tubes of the corresponding fuel element, even in the fully extracted position. The main technical specifications of the vessel are summarized in Table 1.

Table 1 RPV Main Characteristics

Technical Specifications	Unit	Value
Design Pressure	Pa	1.76E7
Design Temperature	°C	350
Inlet normal operating Pressure	Pa	1.611E7
Outlet normal operating Pressure	Pa	1.573E7
Inlet operating Temperature	K	567.65
Outlet operating Temperature	K	598.95
Water Volume	m^3	123.4
Pressure loss across RPV at full load	Pa	2.8E5
Fuel	-	Sintered UO_2
Fuel assemblies	-	177
Fuel assemblies with control assemblies	-	52
Total number of fuel rods	-	41772
Number of guide thimbles per fuel assembly	-	20
Arrangement	-	Square lattice
The overall length of fuel rods	m	4.1850
The active length of one fuel rod at full load	m	3.4146
The outside diameter of fuel rods	m	0.1075

3 NEUTRONIC AND THERMAL-HYDRAULIC MODELS

3.1 PARCS Code: Model Performance and Modifications

PARCS code is a 3D reactor core simulator that can solve the neutron diffusion equation to predict the kinetic response of the reactor against reactivity perturbations such as control rod movements, changes in the fluid temperature, and other conditions on the reactor core. The neutron diffusion equation is solved for two energy groups using the HYBRID method, an Analytical Nodal Method/Nodal Expansion Method (ANM/NEM).

Neutronic and thermal-hydraulic codes can be coupled through the Parallel Virtual Machine (PVM) protocol [10]. To perform the coupling, an input file for PARCS code that specifies the assignment between the nodes of the neutronic model and the nodes of the thermal-hydraulic model is needed. This file, called MAPTAB, is obtained automatically thanks to the tools developed by ISIRYM-UPV with MATLAB® software [11]. The GENINP software obtains PARCS input files automatically [2]. In this way, RELAP5 and TRACE codes will feed PARCS temperature and density distribution information during a transient.

The neutronic model is defined radially and axially. The core is formed by 241 radial nodes and is divided into 34 axial levels. For each fuel assembly a radial node is assigned, so the active zone of the core is composed of 177 nodes, while 64 nodes are for the radial reflector. Between the 34 axial levels, 2 of them correspond to the top and bottom reflectors. Geometric data were provided by CNAT [12].

The simulation requirements for the transient are specified in the PARCS input file. The simulation time is 100 seconds, and the control rod insertion occurs within 50 seconds of the transient with a duration of 2.1 seconds. A null transient of 50 seconds is carried out to ensure stable conditions at the beginning of the simulation. Figure 1 shows the control rod insertion depending on the time (0 cm indicates fully inserted).

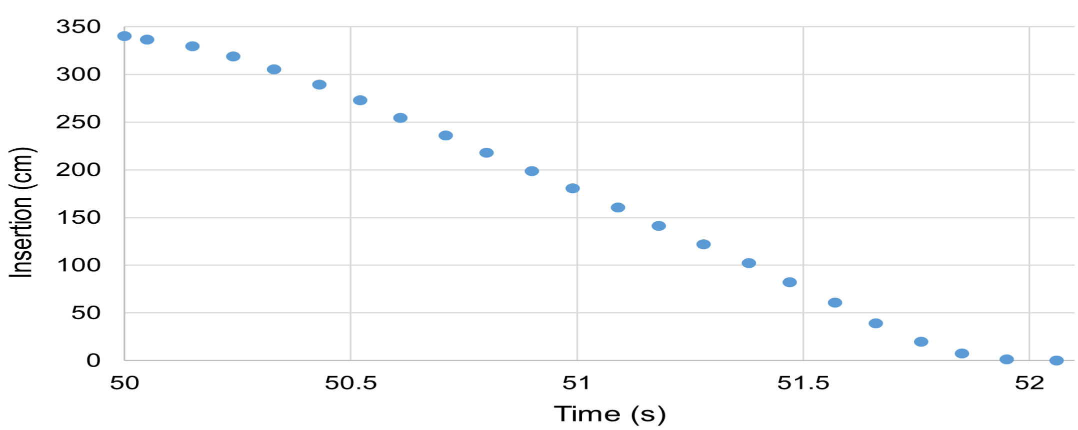

Figure 1 Control Rod Insertion (in cm) Transient

The collaboration between CNAT and ISIRYM-UPV has promoted the study of neutron noise recorded in the Incore and Excore detectors. Simulation of the detector signal has been implemented in the model by changing the source code of PARCS. In this way, the code allows to search for the detectors' position nodes and saves their main variables in separate files, with an acceptable format to optimize the analysis of the results.

3.1.1 Incore Detectors

Incore detectors are n,β-cobalt Power Density Detectors (PDDs). They are distributed in the reactor core in 6 fingers at six axial levels and placed in selected fuel assemblies that constantly measure local neutron flux density. To simulate the response of each PDD, Equation (1) was used [13].

$$P_{ID} = \frac{\Phi}{\Phi_{t=0}} \cdot F_{ID} \tag{1}$$

In this equation, Φ represents the thermal flux while F_{ID} a conversion factor that corresponds to the mean power registered by the PDD at the steady case.

The Incore detectors' position is shown in Figure 2; the control rod inserted is shaded in red.

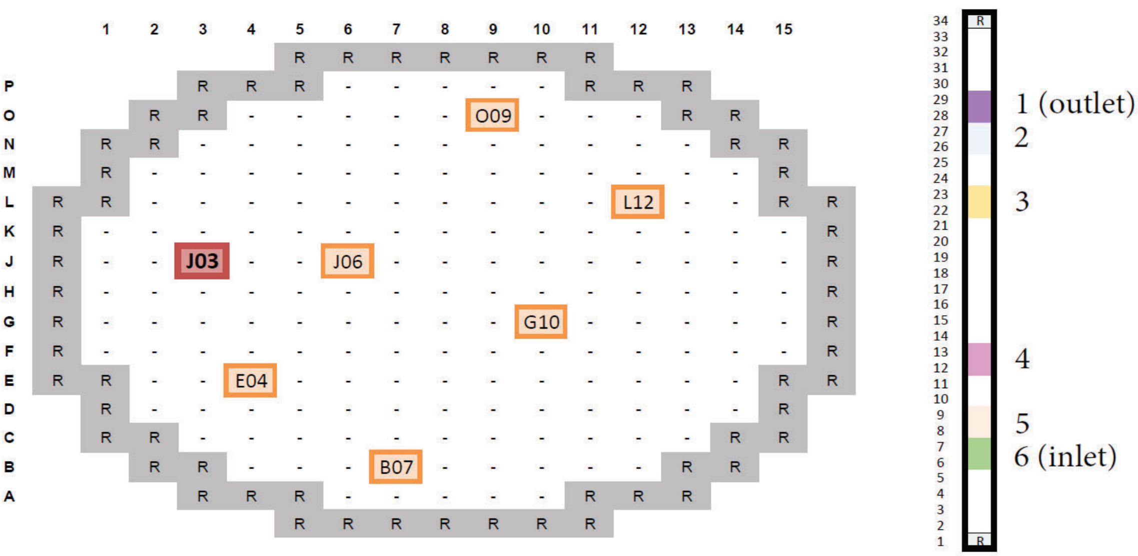

Figure 2 Incore Detectors' Location in the Reactor Core. Shaded, the Control Rod Dropped at the Test

As seen in Figure 2, each PDD finger corresponds to a single radial neutronic node, but the axial position of the Incore detectors in the reactor core is not equivalent to an axial node of the neutronic PARCS model. Table 2 specifies which axial neutronic node of the model corresponds to each axial level of the PDD finger. Therefore, the signal of each PDD in this PARCS model is calculated as the sum of the thermal flux at each axial node multiplied by its corresponding weighting factor. These weighting factors correspond to the actual position of the detector between two axial levels.

Table 2 Axial Weighting Factors for the Incore Detectors

Detector axial level	Actual pos. in nodes	Nodes	Axial weighting fac.
1 (outlet)	28.4	29	0.4
		28	0.6
2	26.6	27	0.6
		26	0.4
3	22.6	23	0.6
		22	0.4
4	12.6	13	0.6
		12	0.4
5	8.3	9	0.3
		8	0.7
6 (inlet)	6.8	7	0.8
		6	0.2

3.1.2 Excore Detectors

The Excore instrumentation is composed of 8 boron-lined ionization chambers. In this case, the detectors are located at the biological shield, out of the core region, which is why a transport model is needed to calculate its simulated signal. In Figure 3, the markings PR1, PR2, PR3, and PR4 indicate four radial positions for the Excore detectors, where each position has two channels, bottom and top.

The *Excore* detectors response is obtained by applying a simple radial transport model. This model is described by Equations (2) and (3) [13]:

$$\Phi(r) \sim \frac{1}{r}\exp(-\Sigma_r r) \qquad (2)$$

$$\Phi(r) \sim \frac{1}{r^2} \qquad (3)$$

On the one hand, Equation (2) describes the neutron transport to the outer vessel surface from each fuel assembly. On the other hand, Equation (3) describes pure geometric transport from the vessel surface to the detectorâ€™s location. The variable r is the distance, and the constant macroscopic cross-section for the medium used in Equation (2) is given in [14], 0.115 cm-1.

This model is applied to the PARCS neutronic nodes â€œseenâ€ by each channel, obtaining the proper Excore weighting factors, as shown in Figure 4. It means that for each Excore location, the nearest nodes contribute to the detector response with a different weight factor depending on the distance.

Axially, each redundancy (top and bottom) is supposed to â€œseeâ€ (detect) half of the core. Then, the axial weighting factor for the bottom detectors is 1 for nodes from 2 to 17 and 0 for nodes from 18 to 33. For top detectors, their axial weighting factors are switched, and the axial weighting factor for nodes 2 to 17 is 0, and nodes from 18 to 33 are 1.

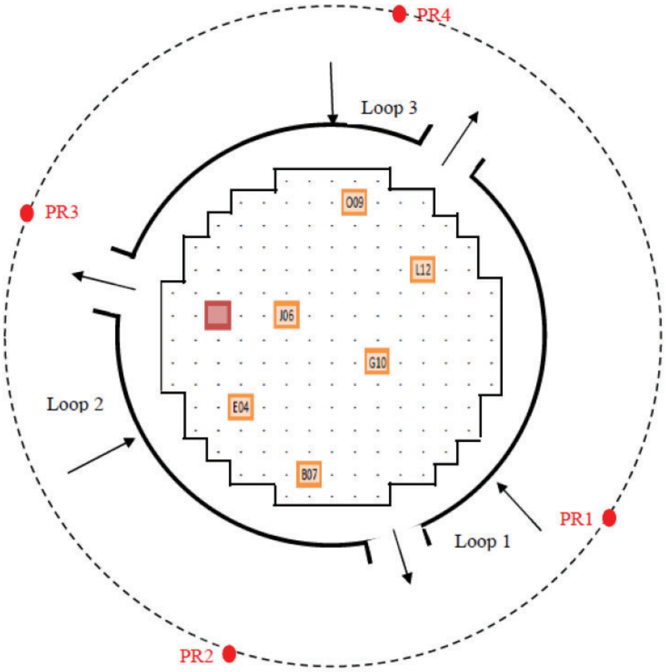

Figure 3 Excore Detectors' Radial Locations

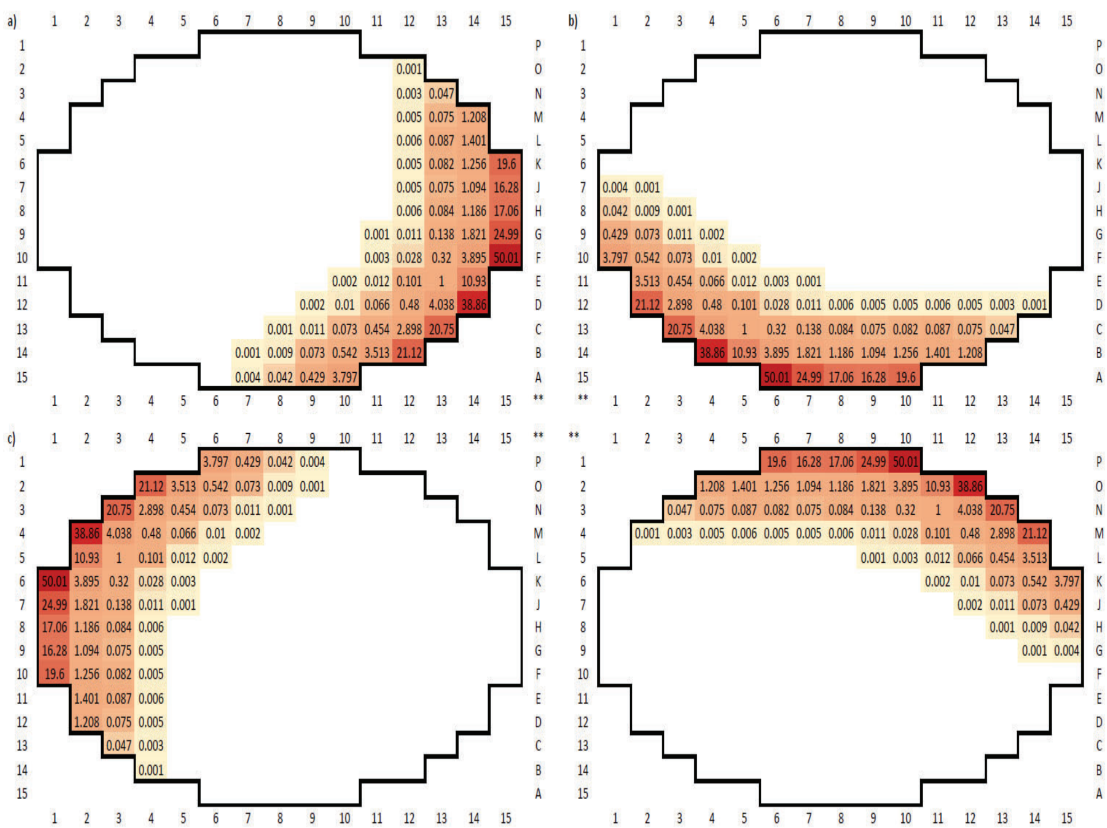

Figure 4 Graphic Representation of the Excore Neutron Detectors Weighting Factors: (a) Exc. 1; (b) Exc. 2; (c) Exc. 3; (d) Exc. 4

3.1.3 PARCS Data Processing

In this work, the neutronic code versions PARCS v2.7 and PARCS v3.0 have been used. RELAP5/mod3.3 has been coupled with PARCS v2.7 since this coupled code is the workhorse of the ISIRYM-UPV research group. The official TRACEv5.0P3 distribution includes coupling with the new neutronic code version, PARCS v3.0. Thus, this distribution is used to test TRACE/PARCS new model options.

In PARCS, subroutine genedits.F was modified to read from external files the information about Incore and Excore detectors, i.e., their location and proper axial and radial weighting factors. Further modifications write the thermal flux and the relative power for the nodes of interest. Then, this data is processed with a MATLAB® program for comparison purposes.

In PARCS, the nodal thermal flux and power are extracted from the regular output file and are processed with MATLAB® generating a proper format for the comparison.

3.2 RELAP5 Thermal-Hydraulic Model

In this study, the thermal-hydraulic model does not contain the reactor coolant loops, so the core mass flow adjustment is carried out by changing the bypass loss coefficients. The model can represent the rod drop transient since this transient has considerable 3D neutron distribution implications. In an actual reactor, the control system would act accordingly to maintain the average temperature. Thus, some control rod banks positions could be modified. Although these are not simulated in this study, these simplifications are accepted for this paper.

A MATLAB program, called CMR5, has been developed at the ISIRYM-UPV to generate RELAP5 thermal-hydraulic models automatically. The data used to prepare this model is obtained from CNAT [15].

The reactor core, with its boundary conditions, is modeled using RELAP5/mod3.3 thermal-hydraulic code. Each fuel assembly is modeled with a pipe + heat structure component (following a one-to-one channel basis), as seen in the sketch obtained with the SNAP tool, Figure 5. The fuel assemblies, which represent the active zone of the core, are divided into 32 axial levels. The same set is used to model the bypass channels.

In the RELAP5 reactor core scheme, the three time-dependent volume + time-dependent junction are used to model the core inlet, corresponding to the three different reactor coolant loops. Inlet boundary conditions such as moderator temperature and mass flow are implemented in these components. These inlets are connected to the channels through three branch components.

As output boundary conditions, the coolant pressure and temperature are defined and implemented in a time-dependent volume component, according to plant data. Coolant pressure will remain constant throughout the transient. The thermal-hydraulic channels are connected to the outlet using three additional branch components corresponding to the three hot legs.

Two thermal-hydraulic radial maps are needed, one input and one output. The input radial map connects each thermal-hydraulic channel to its inlet branch and is based on the coolant inlet loops around the reactor. The theoretical situation of the three bypass elements is also based on the inlet loops situation. The theoretical situation of the three bypass elements is also based on the inlet loops situation. At the same time, the radial output map arranges the channel's outlet in the three superior branch components. Therefore, a channel could be connected to a different inlet/outlet branch (their radial distribution does not spatially match at the inlet and outlet). Both radial maps can be seen in Figure 6.

The plant values used as boundary conditions of the reactor core model are detailed in Table 3.

Table 3 Boundary Conditions Details

Boundary condition	Numerical value
Inlet mass flow loop 1	5578.49 kg/s
Inlet mass flow loop 2	5121.23 kg/s
Inlet mass flow loop 3	5487.04 kg/s
Inlet temperature	567.856 K
Outlet pressure	1.551E7 Pa
Power	3.09E9 W

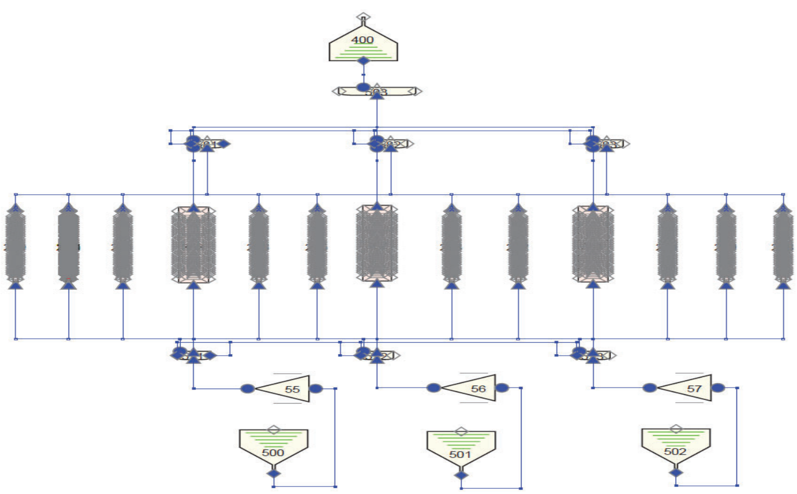

Figure 5 Scheme of RELAP5 Model for the Reactor Core

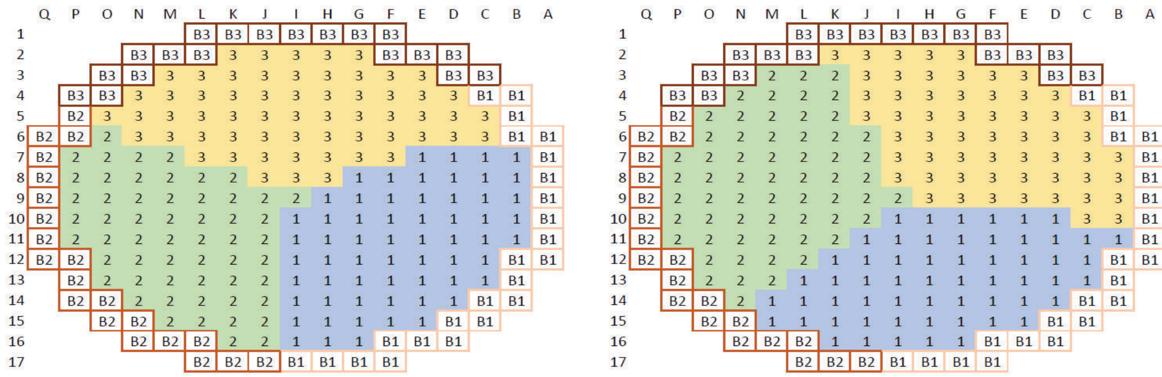

Figure 6 Correspondence between Thermal-Hydraulic Channels and Inlet (on the Left) and Outlet (on the Right) Components Representing the Recirculation Loops

3.3 TRACE Thermal-Hydraulic Model

The TRACE thermal-hydraulic model is based on previous studies [16],[17]. To simulate each fuel assembly, traditional models used pipe or channel components. This new model simulates a realistic, fully 3D core reactor using the 3D vessel component available in TRACE.

A Cartesian vessel is used to model the fuel assemblies. This TRACE component improves previous models providing the cross-flow between each fuel assembly node. Setting to zero the flow area fraction for the corner cells, the Cartesian vessel is shaped as the NPP radial mapping.

The cartesian vessel, which models the reactor core, has been connected to a cylindrical vessel, to reproduce the actual geometry of the reactor pressure vessel. The cylindrical vessel is composed of two radial cells, three azimuthal cells, and 36 axial cells. The inner radial cell represents the bypass, and the outer radial cell represents the downcomer. The three azimuthal cells represent the three different reactor recirculation loops. The lower and the upper axial cells represent the lower and upper reactor plenums. The upper plenum has three various break components, each one connected to a different azimuthal cell. Three fill components are connected to a lower level.

Moreover, one-cell pipes are used in the sideward connections to connect both vessels, and single junctions are used as axial connections. The cylindrical vessel uses one heat structure associated with each bypass azimuthal sector. Finally, one heat structure for each azimuthal sector is used to model the barrel heat transfer between the bypass and the downcomer. The other details, such as assembly heat structures components, azimuthal sector association with assemblies, and bypass, are simulated as was explained for the RELAP5 model. Figure 7 shows the simplified cylindrical model sketch using the SNAP tool (5x5 vessel without lateral junctions).

The core is modeled using 3D components connected from node to node. The bypass mass flow has a substantial variation along its way through the vessel, thus, the bypass friction factor must be adjusted at each node and among the three different azimuthal sectors. An automatic iterative process carries out the adjustment of the bypass friction factor.

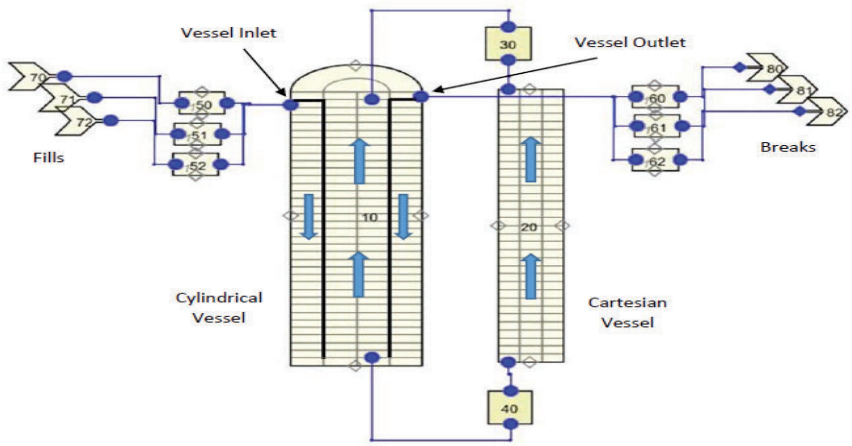

Figure 7 Scheme of TRACE5.0P3 Model for the Reactor Core

The intense effort required to generate the TRACE thermal-hydraulic model, consisting of almost 2600 components and over 150000 input lines, aims to improve accuracy and reproduce a realistic behavior in front of some perturbations, such as different inlet temperatures in each azimuthal zone.

3.4 SIMTAB Methodology for Cross-Sections Acquisition

The Cross-Section data are obtained from CASMO-4/SIMULATE-3 files using the SIMTAB [18] methodology.

SIMTAB methodology was developed at the Institute for Industrial, Radiophysical and Environmental Safety at Universitat Politècnica de València (ISIRYM-UPV) in collaboration with Iberdrola and was validated for both PWR and BWR. Applying this methodology, a set of tabulated cross-sections and kinetic parameters parameterized in terms of local and control variables, such as moderator density, fuel temperature, boron concentration, and control rod, are obtained. In this way, the reactor core is modeled with a few number of neutronic regions that ensure a well-characterized kinetic behavior.

The model is made for fuel cycle 23 of the studied NPP, specifically at the beginning of the cycle (BOC). At this point in the nuclear fuel cycle, applying the SIMTAB methodology, 1379 neutronic compositions have been generated, 1376 for the active core zone and 3 for the lower, upper, and radial reflectors. Thanks to the SIMTAB methodology, neutronic compositions have been reduced from 5600 to 1379.

The generation of cross-sections is verified by the steady-state executions of PARCSv2.7 stand-alone, RELAP5/PARCSv2.7, and TRACE5.0P3/PARCSv3.0. For each simulation, the results of the keff, the absolute error of keff concerning SIMULATE-3, and the root mean square error of the axial and radial power profile are summarized in Table 4. By coupling PARCS with TRACE, keff increases slightly compared to the result of the RELAP5/PARCS simulation. The main reason is the difference in the thermal-hydraulic models between both thermal-hydraulic codes.

Table 4 Results for SIMTAB Cross-Section Validation

CR Dropped/Code	k_{eff}	Abs. error (pcm)*	RMS axial error (%)	RMS radial error (%)
CR J03 PARCSv2.7 stand-alone	1.000812	67.2	3.05	1.88
CR J03 RELAP5/PARCSv2.7	1.000098	46.7	0.82	1.63
CR J03 TRACE5.0P3 /PARCSv3.0	1.001276	113.6	1.36	1.52
*k_{eff} SIMULATE-3 = 1.00014				

Figure 8 represents the radial map of the relative errors of the radial power profile concerning SIMULATE-3, where the results are compared between TRACE/PARCS and RELAP5/PARCS executions. Figure 9 shows the axial power profile for the three simulations performed compared to the SIMULATE-3 reference case.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	TRACE RELAP5					0.00 0.00	0.01 0.01	0.01 0.01	0.00 0.01	0.00 0.00					
2				0.01 0.01	0.00 0.00	0.01 0.01	0.02 0.02	0.01 0.02	0.01 0.01	0.00 0.00	0.01 0.01	0.01 0.01			
3			0.02 0.02	0.02 0.02	0.00 0.01	0.02 0.03	0.02 0.02	0.01 0.02	0.02 0.02	0.02 0.02	0.02 0.03	0.02 0.02	0.02 0.02		
4		0.02 0.02	0.02 0.02	0.00 0.00	0.02 0.02	0.02 0.02	0.01 0.01	0.02 0.02	0.01 0.01	0.02 0.01	0.02 0.02	0.00 0.00	0.02 0.02	0.01 0.01	
5		0.02 0.02	0.03 0.04	0.02 0.02	0.03 0.02	0.02 0.02	0.00 0.00	0.00 0.00	0.00 0.00	0.02 0.02	0.03 0.03	0.02 0.02	0.01 0.01	0.01 0.00	
6	0.02 0.02	0.03 0.03	0.01 0.00	0.01 0.01	0.02 0.01	0.01 0.00	0.00 0.00	0.01 0.01	0.01 0.01	0.01 0.00	0.02 0.01	0.02 0.01	0.02 0.02	0.01 0.01	0.01 0.00
7	0.01 0.01	0.02 0.02	0.00 0.01	0.00 0.00	0.00 0.00	0.01 0.01	0.03 0.03	0.01 0.01	0.03 0.03	0.00 0.00	0.00 0.01	0.01 0.01	0.01 0.01	0.01 0.01	0.00 0.00
8	0.01 0.01	0.02 0.02	0.00 0.00	0.03 0.03	0.00 0.00	0.01 0.01	0.01 0.01	0.02 0.02	0.01 0.01	0.01 0.01	0.00 0.00	0.03 0.03	0.00 0.00	0.02 0.01	0.01 0.01
9	0.00 0.00	0.01 0.01	0.01 0.01	0.01 0.01	0.00 0.00	0.00 0.00	0.03 0.03	0.01 0.01	0.03 0.03	0.01 0.01	0.00 0.00	0.00 0.00	0.00 0.01	0.02 0.02	0.01 0.01
10	0.01 0.00	0.01 0.00	0.02 0.02	0.02 0.01	0.02 0.02	0.01 0.00	0.01 0.01	0.01 0.01	0.00 0.00	0.01 0.00	0.02 0.02	0.01 0.01	0.01 0.01	0.03 0.03	0.02 0.02
11		0.01 0.00	0.01 0.01	0.02 0.02	0.03 0.03	0.02 0.02	0.00 0.00	0.00 0.01	0.00 0.00	0.02 0.02	0.03 0.03	0.02 0.02	0.03 0.04	0.01 0.01	
12		0.01 0.01	0.02 0.02	0.00 0.00	0.02 0.02	0.02 0.02	0.01 0.01	0.02 0.01	0.01 0.01	0.02 0.02	0.02 0.03	0.00 0.00	0.02 0.02	0.02 0.02	
13			0.02 0.02	0.02 0.02	0.02 0.03	0.02 0.02	0.02 0.02	0.01 0.02	0.02 0.02	0.02 0.03	0.01 0.00	0.02 0.02	0.02 0.02		
14				0.02 0.01	0.01 0.01	0.00 0.00	0.01 0.01	0.01 0.02	0.02 0.02	0.01 0.01	0.00 0.00	0.01 0.01			
15						0.00 0.00	0.00 0.01	0.01 0.01	0.01 0.01	0.00 0.00					

Figure 8 Relative Radial Errors Concerning the Reference, SIMULATE-3. Comparison between RELAP5 and TRACE Coupled Steady State Case Results

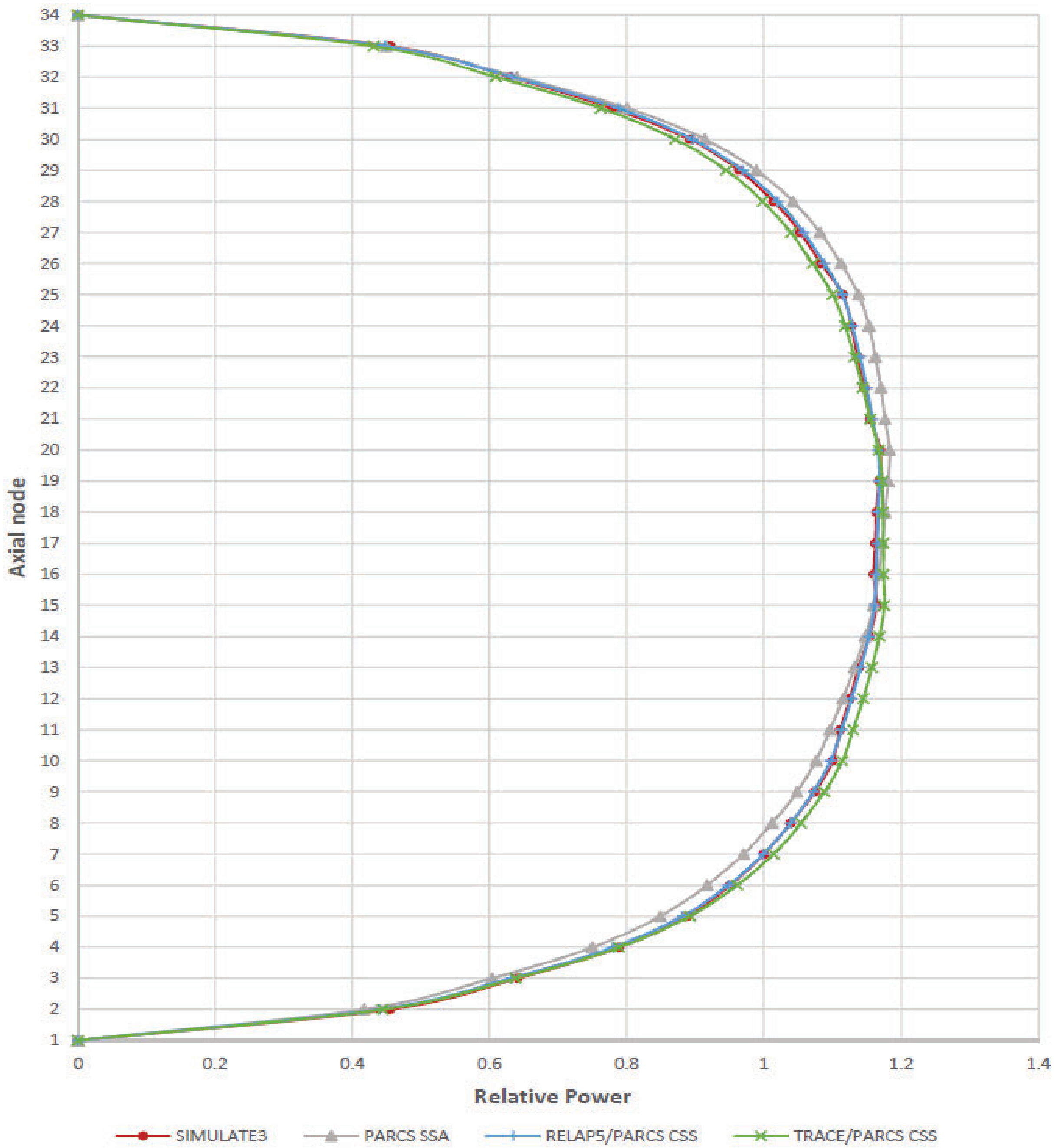

Figure 9 Relative Power per Axial Plane. Comparison between PARCS Stand-Alone Results, Coupled Stationary Case Results for RELAP5 and TRACE Codes, and the Reference, SIMULATE-3

4 RESULTS

The results of the transient executed with the coupled codes RELAP5/PARCSv2.7 and TRACE5.0P3/PARCSv3.0 are presented in the following.

The total power evolution during the transient is very similar for both codes, as seen in Figure 10.

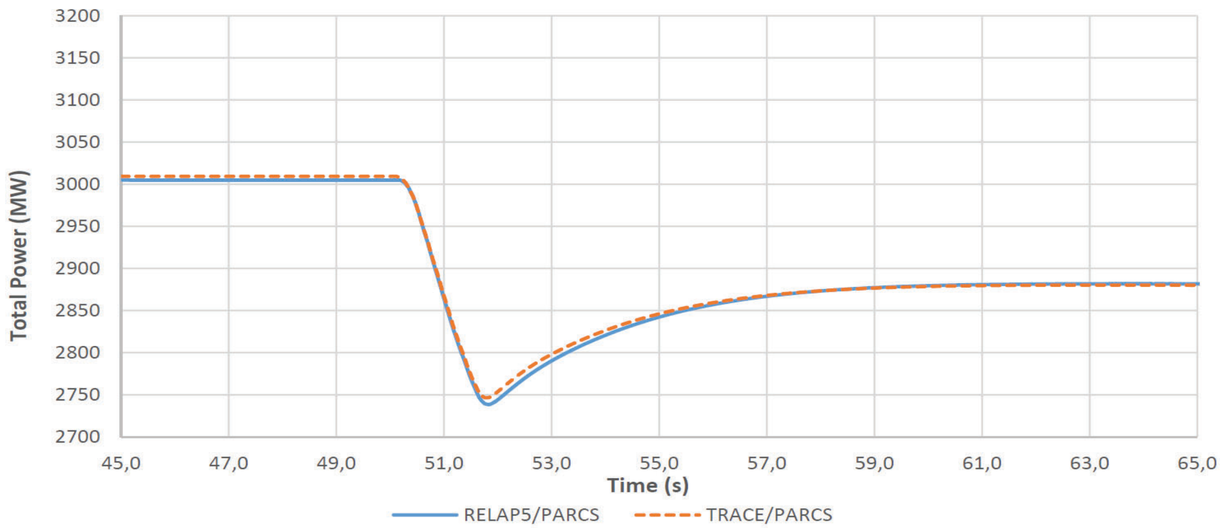

Figure 10 Total Power Evolution. Comparison between RELAP5/PARCS and TRACE/PARCS Results

The control rod drop causes the insertion of negative reactivity, that decreases the nuclear power and, consequently, the moderator temperature. As the moderator temperature decreases, its density grows, causing an increase in the moderation of neutrons in the reactor core. Moderation of neutrons produces a growth in the reaction rate, which results in an increase in fuel temperature and reactor power. A new stationary point is reached within a few seconds.

The negative reactivity caused by the control rod drop is counteracted by the thermal-hydraulic feedback effect introduced by the moderator and fuel temperature reactivities, as shown in Figure 11. Moderator temperature decreases as the nuclear power decreases (by the absorber insertion). Consequently, the moderator density grows, causing an increase in the moderation of neutrons in the reactor core. This produces a growth in the reactions rate, so in a few seconds, the fuel temperature and the power increase until a new stationary point is reached. The evolution of maximum moderator, Doppler and fuel temperatures has been shown in Figure 12.

As for the coolant pressure, the outlet pressure is fixed as a boundary condition in the model. Figure 13 shows how the inlet/outlet coolant pressures obtained with RELAP5 and TRACE thermal-hydraulic codes remain constant.

During the 50-second null transient, the inlet and outlet coolant pressures are 100%. When the control rod is inserted, the coolant inlet pressure decreases for both codes. This pressure evolution proves that the acceptance criteria mentioned in the introduction are fulfilled.

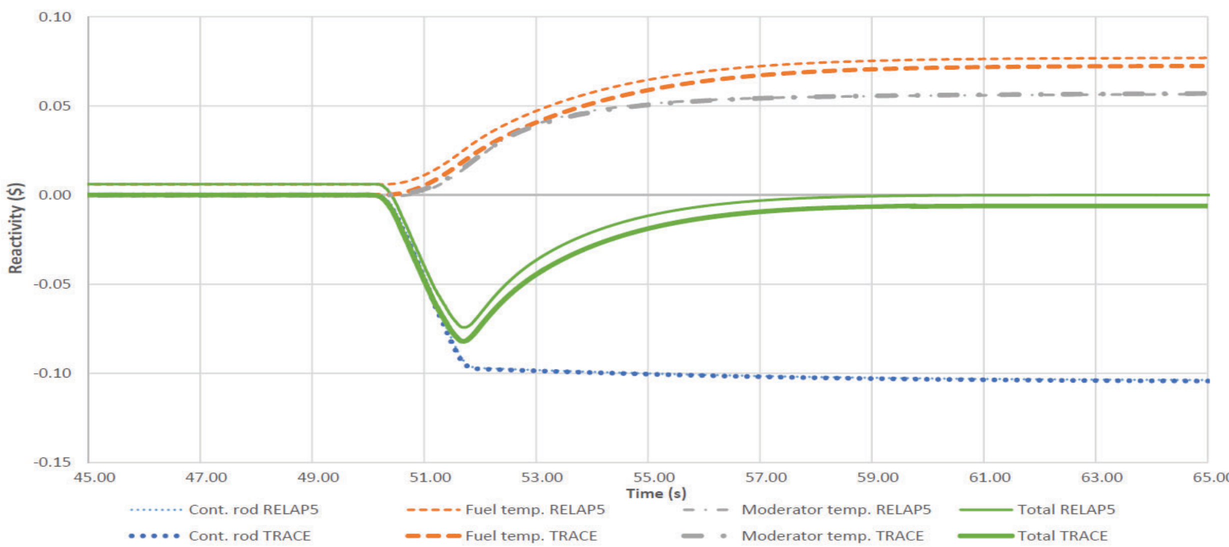

Figure 11 Reactivity Evolution

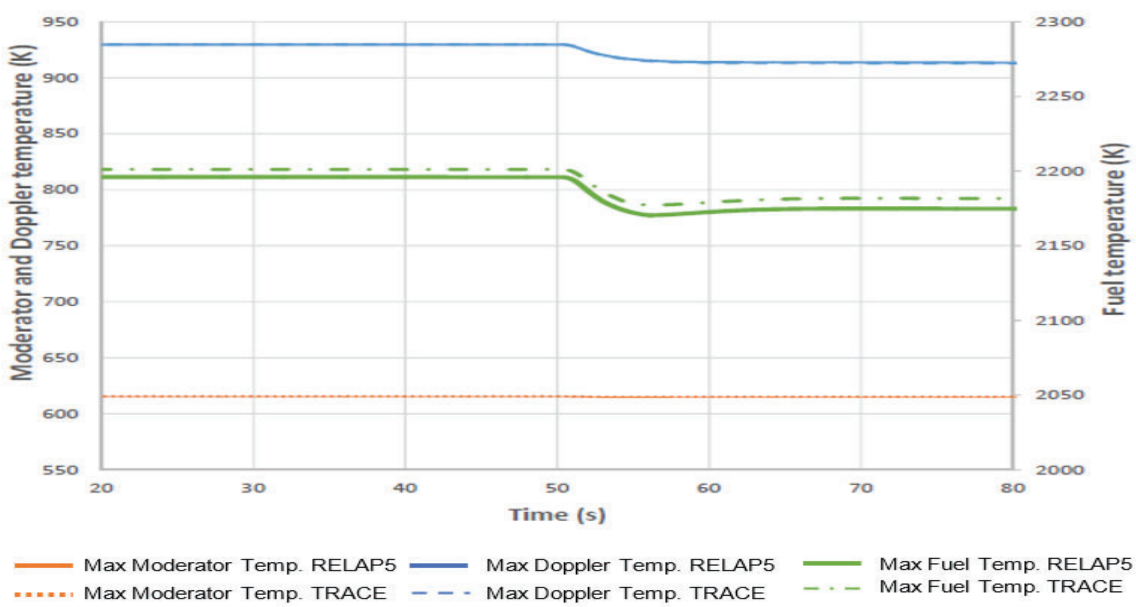

Figure 12 Temperatures Evolution

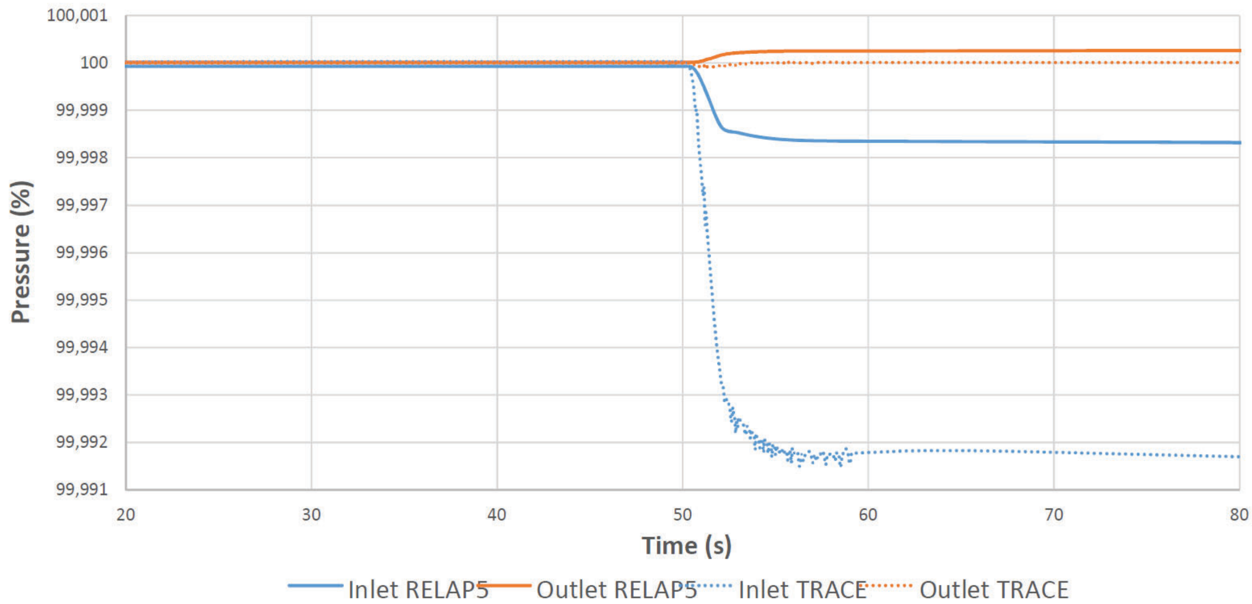

Figure 13 Pressure Evolution

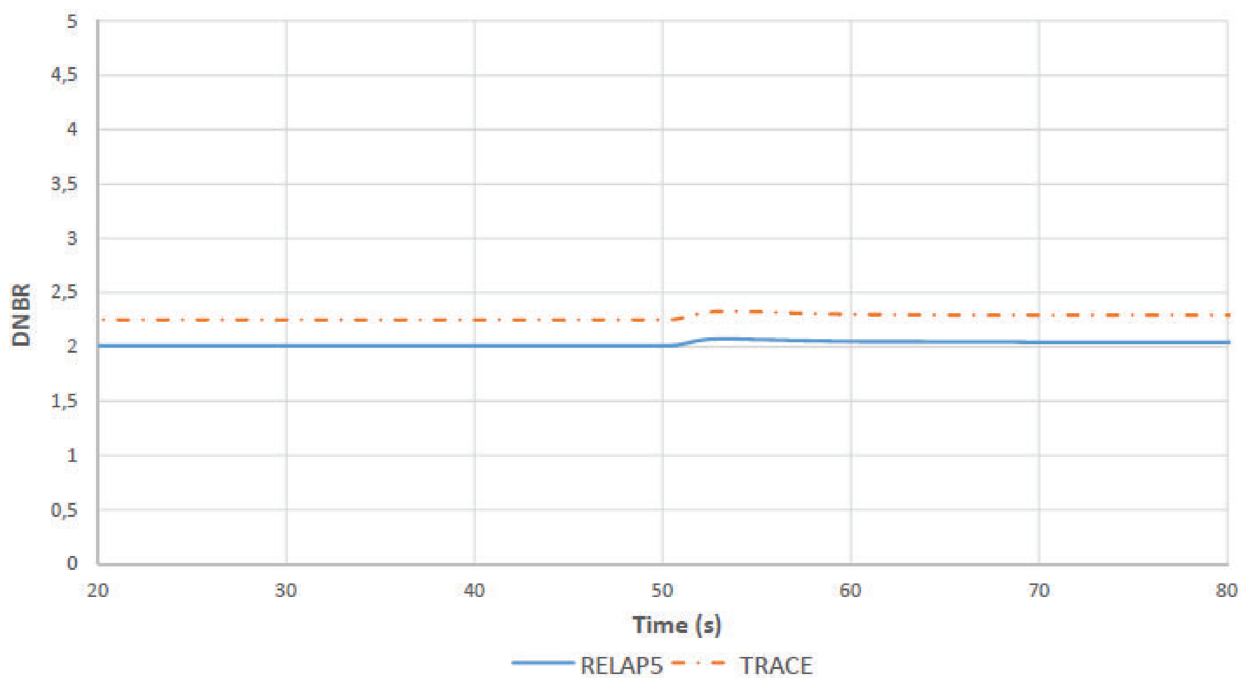

Figure 14 Departure from Nucleate Boiling Ratio (DNBR) Evolution

Regarding the DNBR calculation, Figure 14 shows that RELAP5 and TRACE codes use different methods to obtain the Critical Heat Flux (CHF). RELAP5 [19] uses the 1986 AECL-UO Critical Heat Flux Lookup Table method [20]. This table is obtained for tubes with a 0.008 m diameter. Up to eight multiplying factors are applied to correct the CHF values obtained from the tables. TRACE [21] uses the AECL-IPPE CHF Table [22] and implements only two multiplying factors to correct the CHF values obtained from the tables (previous version [20]). Besides, these two factors are used in an exclusionary manner, and therefore only one correction is applied to the CHF obtained in TRACE. These factors are the correction for tube diameter, k1, and rod bundle geometry, k2. TRACE considers that rod bundle geometry is always used (pitch-to-diameter ratio always greater than one). Therefore, TRACE will always apply only the k2 correction factor. However, the difference between the actual hydraulic diameter and the diameter in the tables is not negligible. Thus, the TRACE code was modified to apply both factors to obtain the CHF final value.

The control rod dropped is located at the J03 radial position of the reactor core. In Figure 3, the detectors closest to the control rod can be observed: J06 and E04 Incore detectors and Excore channels PR3.

The signal analysis of the detectors is performed by comparing the actual signals provided by the CNAT [23], with the results obtained by the coupled codes and by SIMULATE-3K code, which is an advanced, two-group nodal code that delivers neutronic and thermal-hydraulic analysis with licensing-grade accuracy. In the Excore detectors, the signals at the bottom and top channels are analyzed. At the same time, the signals at axial levels 6 (inlet) and 1 (outlet) are examined of the six axial levels that form a PDD. Fluctuations observed on the actual data signals for all detectors are due to neutronic noise.

The Incore detectors detect neutron flux signals at the reactor core position at the nodes where they are inserted. First, the linear power density evolution at PDD J06, the closest to the fallen control rod, is analyzed. The coupled codes obtain a similar evolution, Figures 15 and 16. If we compare their results with the SIMULATE-3K data, the maximum relative error observed is around 0.25% for the minimum power reached. For the steady-state operating point reached after the control rod drop, the maximum error is about 1.17%. It has been observed that these maximum errors have been obtained with the RELAP/PARCS code. The signals from PDD E04, the second closest Incore detector to the dropped control rod, have also been analyzed in Figures 17 and 18. The results obtained with the TRACE/PARCS code are closer to the linear power density evolution provided by the SIMULATE-3K code. Therefore, the maximum relative errors are obtained in the transient simulation with RELAP/PARCS. For the minimum power reached, the maximum relative error observed is around 0.5%, while for the steady-state operating point reached after the control rod drop, the maximum error is about 0.9%.

The signal detected by Excore detectors corresponds to the average of a significant number of nodes, about 60 radial positions and 16 axial positions. Excore channels PR3 are out of the core, so the signal is decreased by a geometrical transport equation. Figures 19 and 20 show the evolution of the power in the top and bottom channels of the Excore detector located at the PR3 position. The results obtained are very close to the actual plant data, although the power drop in the top channel is faster in the signals calculated by the codes. Comparing their results with SIMULATE-3K data, for the minimum power reached, the maximum relative error observed is around 2.0% for the TRACE/PARCS simulation. For the steady-state operating point reached after the control bus drop, the maximum error is about 1.3%, and in this case, it is obtained with the RELAP/PARCS code.

The results of the signals of all Incore and Excore detectors implemented in the PARCS neutron model are compared concerning SIMULATE-3K. Figures 21 and 22 summarize the difference in % between the SIMULATE-3K data and the results calculated by the coupled RELAP/PARCS and TRACE/PARCS codes. Figure 21 shows the values for the minimum value of the simulated signal after the control bus drop. In contrast, Figure 22 shows the value of the simulated signal after reaching the new power level.

The relative errors in the evolution of the detector signal between the coupled thermalhydraulic-neutronic codes and the actual signals have also been calculated. For the minimum power reached during the control rod drop, the maximum relative error is about 5% for all three codes compared (RELAP5/mod3.3/PARCSv2.7, TRACE5/PARCSv3.0, and SIMULATE-3K). The maximum relative error for the steady-state operating point reached after the control rod drop is around 2.5% for all three codes.

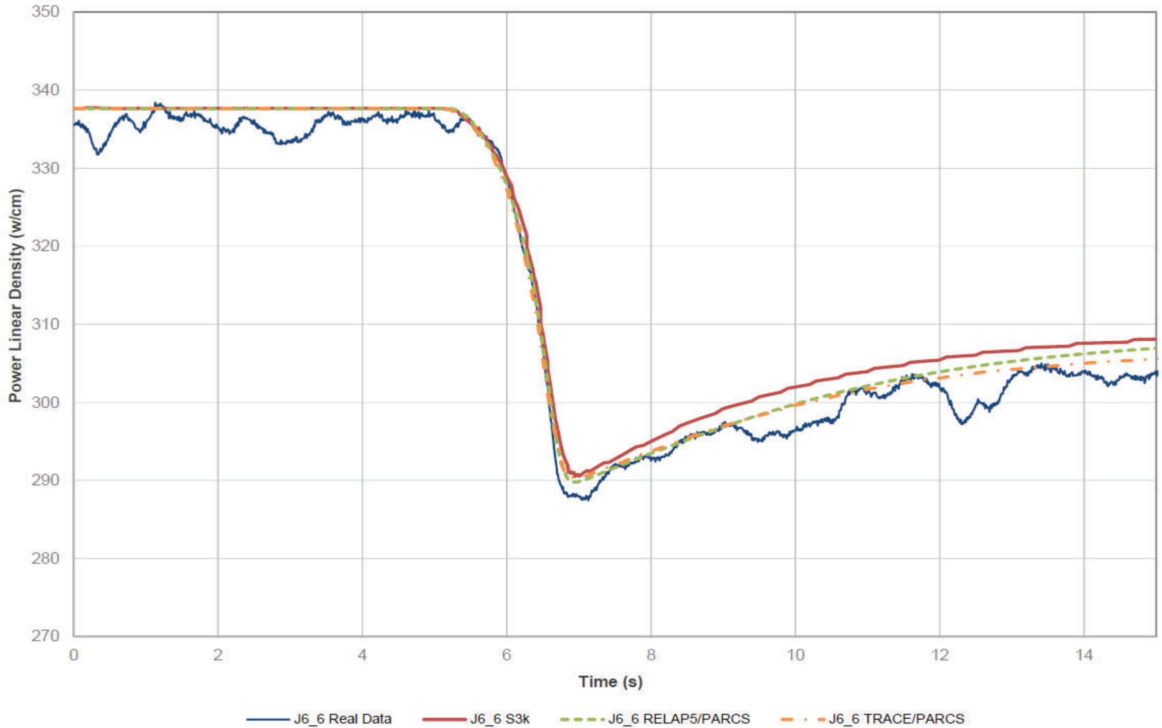

Figure 15 Real and Calculated Signals for Incore Detector J06, Axial Level 6 (Inlet)

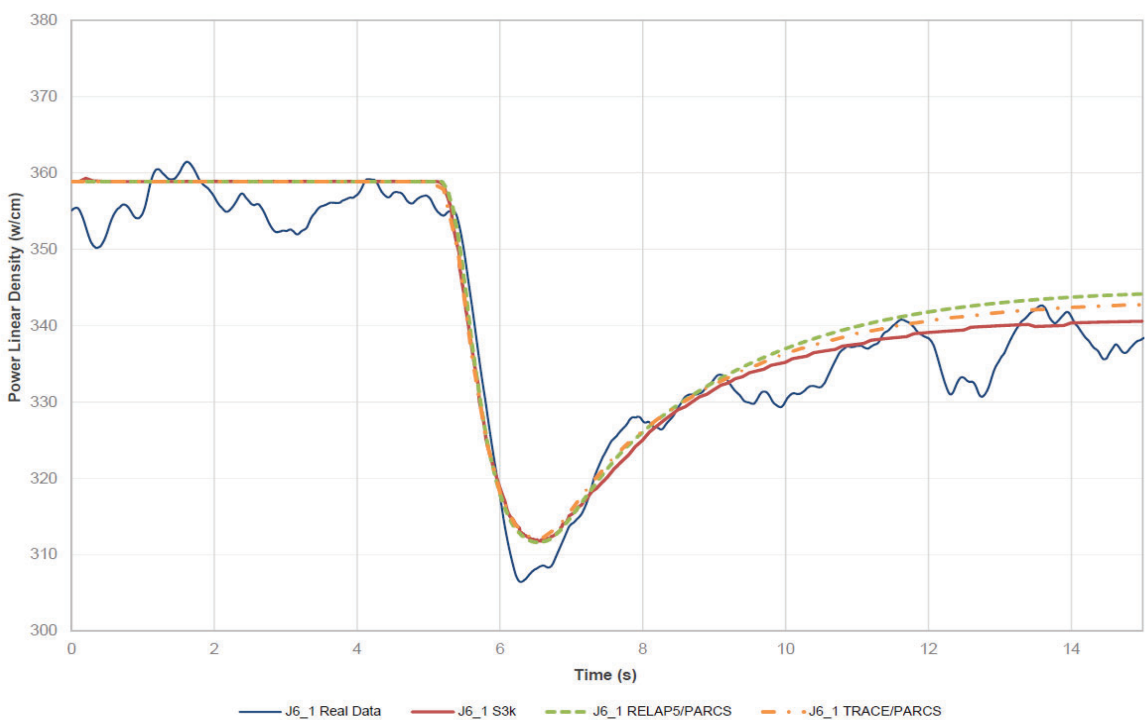

Figure 16 Real and Calculated Signals for Incore Detector J06, Axial Level 1 (Outlet)

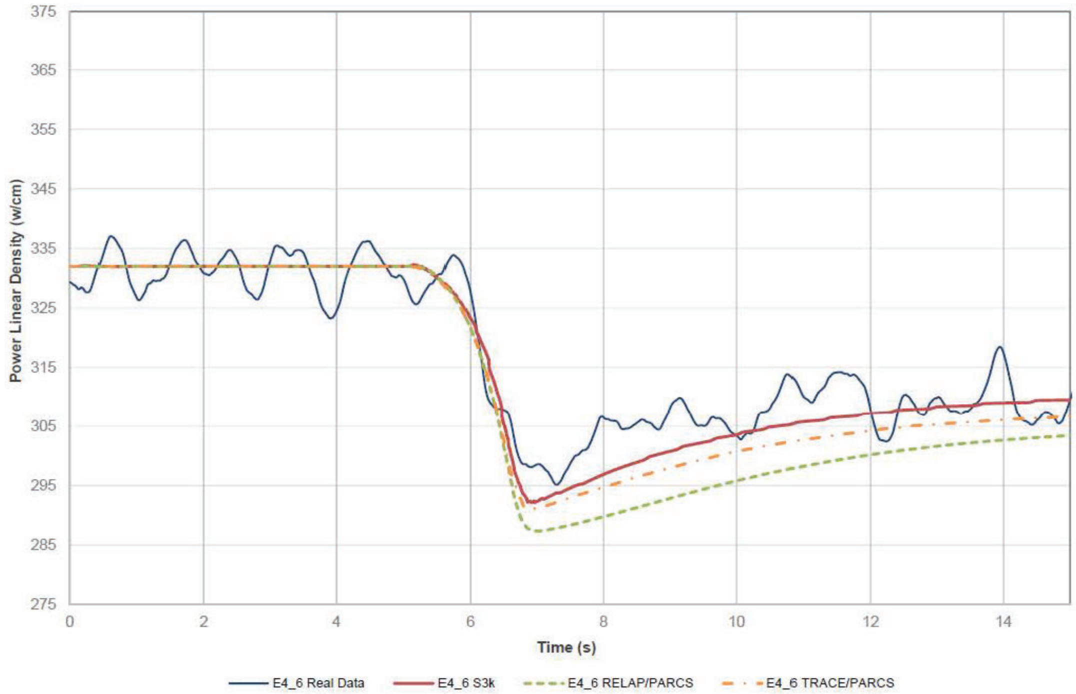

Figure 17 Real and Calculated Signals for Incore Detector E04, Axial Level 6 (Inlet)

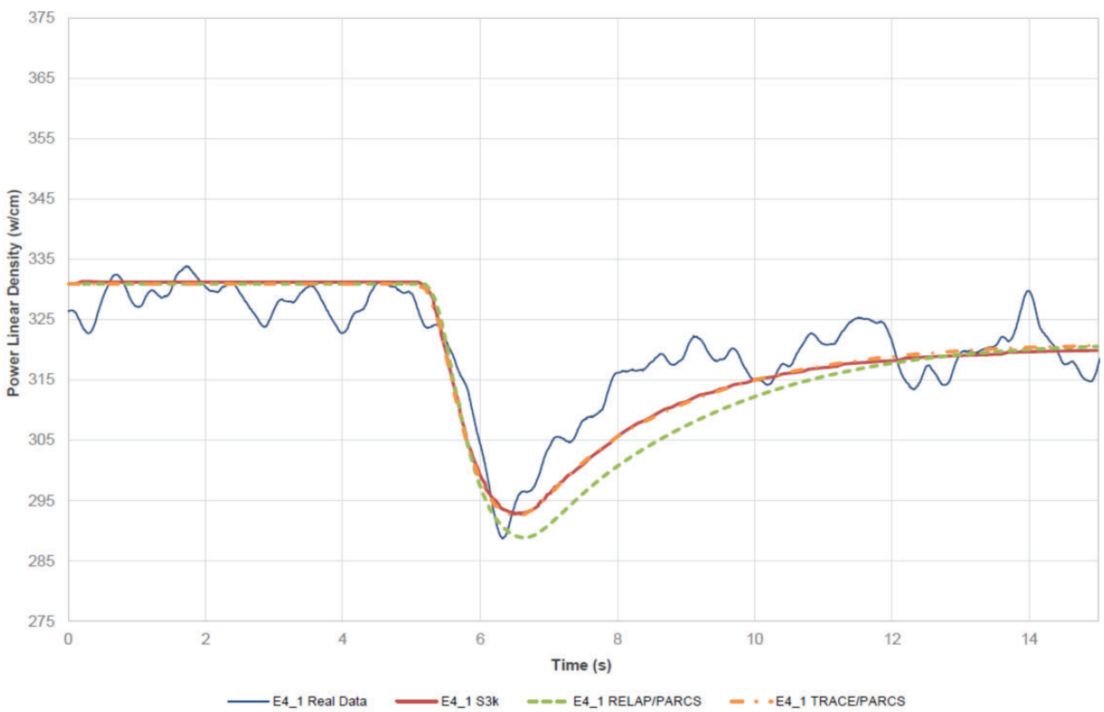

Figure 18 Real and Calculated Signals for Incore Detector E04, Axial Level 1 (Outlet)

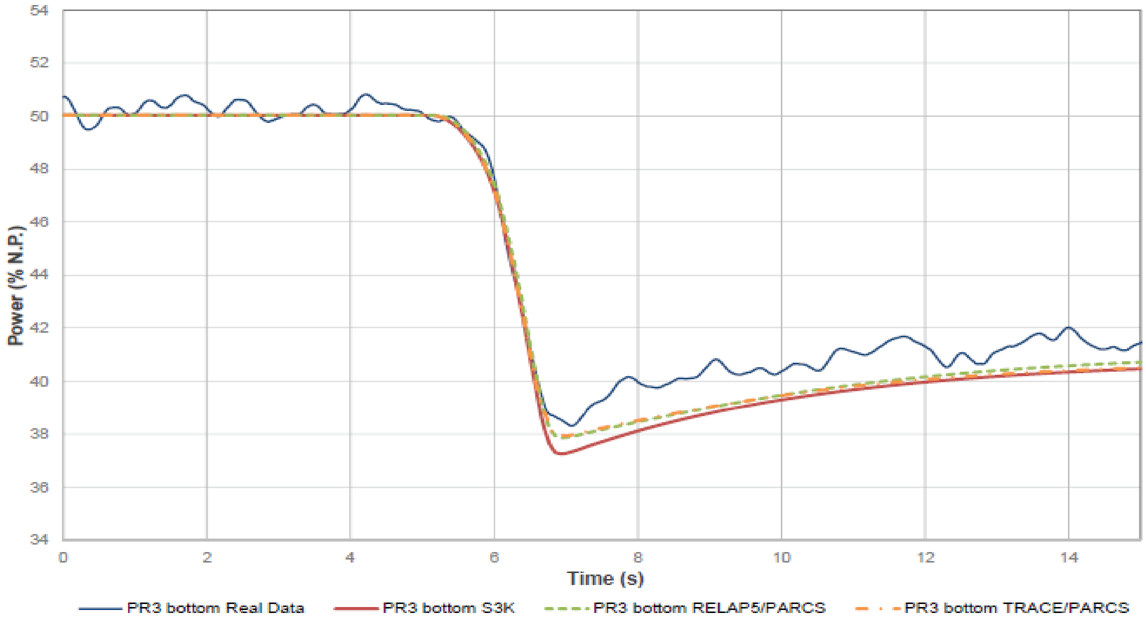

Figure 19 Real and Calculated Signals for Excore Detector PR3, Bottom

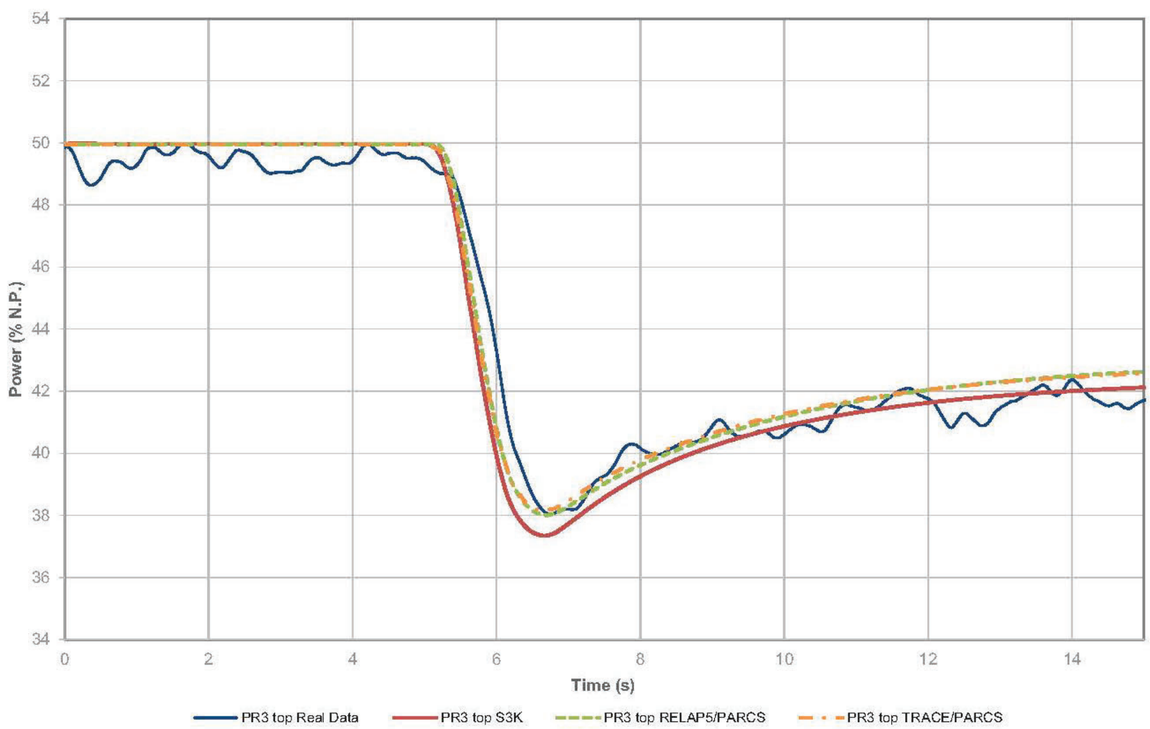

Figure 20 Real and Calculated Signals for Excore Detector PR3, Top

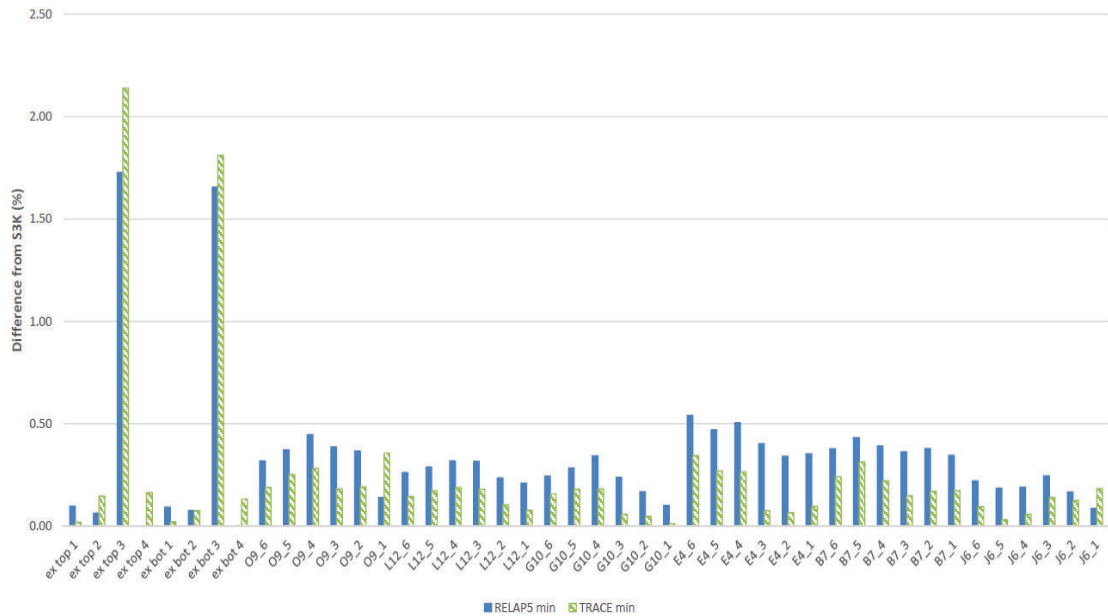

Figure 21 The Difference, in %, between RELAP5 and TRACE Codes Results and SIMULATE-3K Results. The Minimum Value of the Simulated Signal After the Control Rod Drop for Incore and Excore Detectors

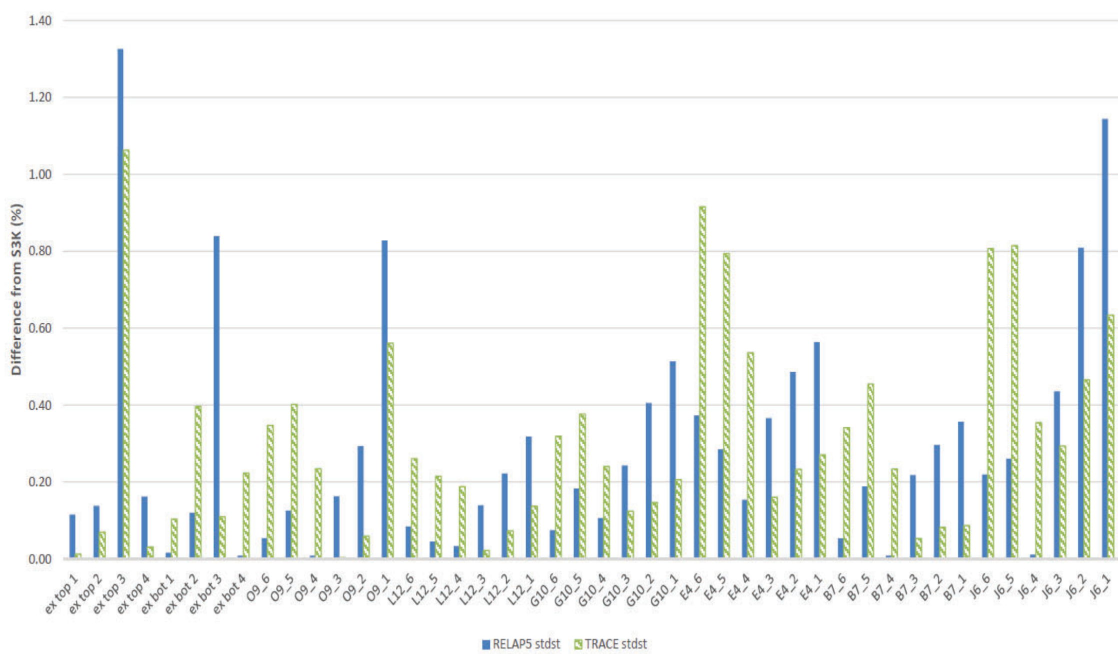

Figure 22 The Difference, in %, between RELAP5 and TRACE Codes Results and SIMULATE-3K Results. Simulated Signal Value once the New Power Level is Reached for Incore and Excore Detectors

Finally, the 3D cross-flow predicted by TRACE5/PARCS is evaluated in axial level 15 near the control rod inserted. The 3D TRACE thermal-hydraulic model defines the reactor core by a cartesian vessel formed by a radial mesh of 15x15 cells (X and Y axis) and 36 axial levels (Z axis). Figure 23 shows the cross-flow change (in kg/s) among the node where the control rod is inserted and its neighboring nodes during the control rod drop transient. These nodes are represented in a scheme in Figure 24, where the shaded square is the radial cell where the control rod is inserted. This diagram also shows the nomenclature of each surface, X or Y, through which the cross-flow flows.

The control rod insertion causes a local decrease in the moderator temperature, thus, the moderator density increases and the pressure decreases. This differential pressure provokes increments on the cross-flows flowing from the nearer nodes to the node where the control rod is inserted, as seen in Figure 23.

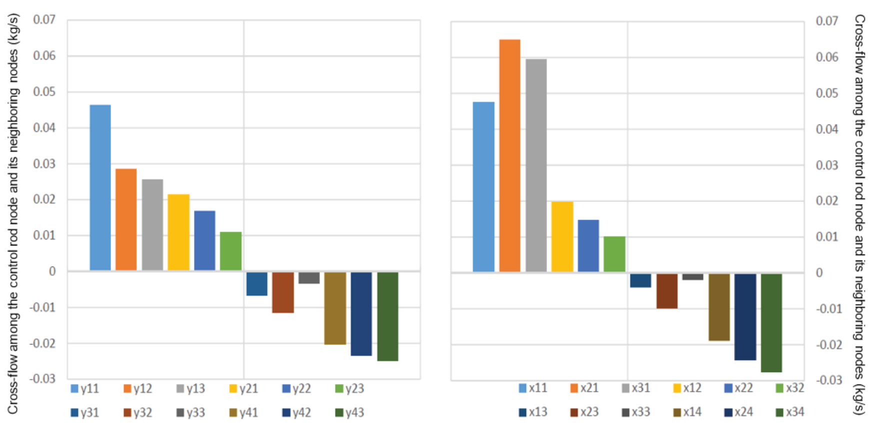

Figure 23 Cross-Flow Change During the Transient in the Y-Axis (Right) and X-Axis (Left)

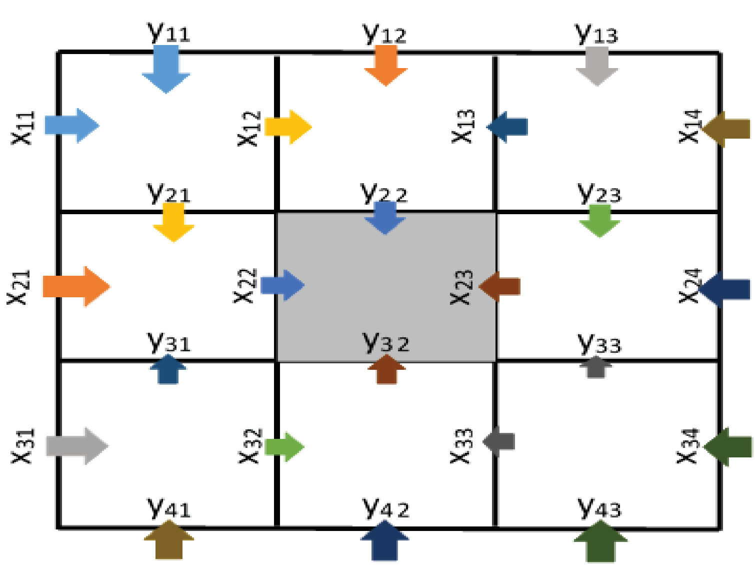

Figure 24 Scheme of the Analyzed Cross-Flows

5 COMPUTING STATISTICS

The control rod drop transient simulation was made using two thermalhydraulic-neutronic coupled codes: RELAP5/PARCSv2.7 and TRACE5.0P3/PARCSv3.0. The simulation has been run in a high-performance computing cluster called Quasar which has 7 nodes with 32 AMD Opteron© CPU each and 1 node with 24 CPUs.

Before reproducing the control rod drop transient, the boundary conditions in the thermal-hydraulic models are adjusted by running RELAP5 and TRACE stand-alone steady-state. The generation of cross-sections is validated by running PARCS stand-alone steady-state, and RELAP5/PARCSv2.7 and TRACE5.0P3/PARCSv3.0 coupled steady-state. Therefore, the total CPU time is the sum of the four simulations. In this event, PARCS stand-alone simulation, for both 2.7 and 3.0 versions, is not considered since the CPU time required is about 3 seconds, which is deemed negligible for this comparison. Table 5 compares each code's computational cost and shows the simulation's total CPU time. It can be observed that the high definition obtained with TRACE code has a crucial computational cost compared to RELAP5 code performance.

Table 5 Total CPU Time (s)

SIMULATION CASE	RELAP5/PARCSv2.7	TRACE5.0P3/PARCSv3.0
Stand-alone steady-state	2719.8 s	3051.0 s
Coupled steady-state	4606.6 s	834.0 s
Coupled transient	4356.3 s	27067.0 s
Total CPU time	11682.7 s	30952.0 s

6 CONCLUSIONS

This study aims to analyze the ability of the coupled TRACE/PARCS and RELAP5/PARCS codes to reproduce a Control Rod Drop Transient in a KWU-PWR nuclear power plant.

To carry out this study, two different thermal-hydraulic-neutronic models have been generated to reproduce the operational transient and to compare their results with the data recorded by the plant during the control rod drop. During this test, a considerable amount of data was recorded, including the actual signals from the Incore and Excore detectors. In addition, the plant also provides the signal results of the detectors by simulating the transient with the SIMULATE-3K code.

Simulation of the Incore and Excore detector signals has been implemented in the model by modifying the PARCS source code. PARCS neutronic code can be coupled with thermal-hydraulic codes such as RELAP5 and TRACE. To perform the coupling, an input file is needed for the PARCS code that specifies the mapping between the neutronic model nodes and the thermohydraulic model nodes. This file, called MAPTAB, is obtained automatically thanks to the tools developed by ISIRYM-UPV with MATLAB® software. In the case of coupling with the RELAP5 code, it is also necessary to use the Parallel Virtual Machine (PVM) protocol.

The thermal-hydraulic model generated with the RELAP5 code is a reactor core model, while the TRACE thermal-hydraulic model is a 3D model of a reactor pressure vessel. A cartesian vessel component is used to model the fuel assemblies. This TRACE component improves on previous models providing the cross-flow between each fuel assembly node. In this model, cross-flow predicted by TRACE5/PARCS at the axial level 15 near the inserted control rod is evaluated.

Cross-Section generation is verified using steady-state runs of PARCS stand-alone, RELAP5/PARCSv2.7, and TRACE5.0P3/PARCSv3.0. For each simulation, the absolute keff error and the root mean square error of the axial and radial power profile are analyzed relative to SIMULATE-3K results. For the coupled steady-state runs, the RMS error of the axial and radial power profile is less than 2%, and the value of the absolute keff error in the PARCS/TRACE increases slightly to the RELAP5/PARCS simulation result, from 46.7 to 113.6 pcm. Therefore, the results show good agreement with the SIMULATE-3K reference code.

The Control Rod Drop Transient is reproduced using the coupled codes RELAP5mod3.3 and PARCSv2.7 and TRACEv5.0P3/PARCSv3.0. The total simulation time is 100 seconds, and after a zero transient of 50 seconds, the control rod is inserted in 2.1 seconds.

These 100 seconds of transient, reproduced on a high-performance computing cluster, correspond to a CPU time of 4356.3 s for the RELAP/PARCS code and 27067.0 s for the TRACE/PARCS code. Adding the time of the above steady-state simulations, the total computational cost for each coupled code, RELAP5/PARCSv2.7 is 11682.7 and TRACE5.0P3/PARCSv3.0, is 30952.0 s, respectively. It can be observed that the high definition obtained with TRACE code has a crucial computational cost compared to RELAP5 code performance.

Using different thermal-hydraulic models, the results obtained in RELAP5 and TRACE have been compared. In particular, the following variables are analyzed: the evolution of the total reactor power, the reactivity introduced by the control rod drop, the reactivity introduced by the moderator and fuel temperature, the evolution of the maximum fuel and moderator temperature, the evolution of the pressure and the Departure from Nucleate Boiling Ratio evolution. Also, the 3D cross-flow obtained by TRACE5/PARCS at the axial level 15 near the inserted control rod is evaluated.

Signals from the Incore and Excore detectors closest to the dropped control rod are analyzed in comparison with the actual data recorded in the plant and the results obtained with the SIMULATE-3K reference code. The control rod dropped is located at the J03 radial position of the reactor core, therefore, the detectors closest to the control rod are: J06 and E04 Incore detectors and the channels PR3 of the Excore detectors.

Since the results obtained are very close to the actual data provided by the plant, this study provides an example of the ability of coupled codes to simulate operational transients, such as a control rod drop. Thanks to all the experimental data provided by CNAT, it has been possible to perform a qualification of the 3D thermalhydraulic-neutronic codes used to reproduce this transient; at the same time that new valuable tools have been developed for future studies.

For the coupled codes RELAP5/PARCSv2.7 and TRACE5.0P3/PARCSv3.0, the ability to obtain the response of Incore and Excore detectors is essential in neutron noise studies, has been validated. Future works will include the simulation of different perturbations of the thermal-hydraulic inlet parameters. Since these models describe three inlet loops with separated components, introducing perturbations in each loop will be easily accomplished. Furthermore, given the multiple possibilities for the cartesian vessel TRACv5.0P3 model will be modified to improve its capabilities.

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

A procedure to analyze the PWR response to local deviations on neutronic or thermal-hydraulic parameters is being developed. By neutronic-thermalhydraulic coupled codes, Incore and Excore neutron flux detector signals are simulated. These signals are compared, on the one hand, with the actual data collected during a control rod drop test at a PWR NPP and, on the other hand, with data obtained with SIMULATE-3K code, an advanced, two-group nodal code that delivers neutronic and thermal-hydraulic analysis with licensing-grade accuracy. At the same time, the used codes and their new capabilities are validated. The 3D neutronic-thermalhydraulic codes used in this study are RELAP5/PARCSv2.7 and TRACEv5.0P3/PARCSv3.0. TRACEv5.0P3 and RELAP5 thermal-hydraulic models are full-core detailed models with three different azimuthal zones. Besides, the TRACE model is performed with a fully 3D core composed of a cartesian vessel representing the fuel assemblies and a cylindrical vessel representing the bypass and downcomer. Cross-section data are obtained from CASMO-4/SIMULATE-3 files using the SIMTAB methodology, which was developed at the Institute for Industrial, Radiophysical and Environmental Safety at Universitat Politècnica de València (ISIRYM-UPV) in collaboration with Iberdrola and has been validated for both PWR and BWR.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.) NPP, PWR, Control Rod, Neutronic, Thermal-Hydraulic, Incore and Excore Neutron Flux, RELAP5, PARCS, TRACE, SIMULATE-3K, CASMO-4, Cross-section Data	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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